INEEL/CON-02-00442
PREPRINT

ATR WG-MOX Fuel Pellet Burnup Measurement By Monte Carlo – Mass Spectrometric Method

G. S. Chang

October 7, 2002

PHYSOR 2002

ATR WG-MOX FUEL PELLET BURNUP MEASUREMENT BY MONTE CARLO – MASS SPECTROMETRIC METHOD

G.S. Chang
Idaho National Engineering & Environmental Laboratory
P. O. Box 1625
Idaho Falls, ID 83415-3885
e-mail. gsc@inel.gov

ABSTRACT

This paper presents a new method for calculating the burnup of nuclear reactor fuel, the MCWO-MS method, and describes its application to an experiment currently in progress to assess the suitability for use in light-water reactors of Mixed-OXide (MOX) fuel that contains plutonium derived from excess nuclear weapons material.

To demonstrate that the available experience base with Reactor-Grade Mixed uranium-plutonium OXide (RG-MOX) can be applied to Weapons-Grade (WG)-MOX in light water reactors, and to support potential licensing of MOX fuel made from weapons-grade plutonium and depleted uranium for use in United States reactors, an experiment containing WG-MOX fuel is being irradiated in the Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory.

Fuel burnup is an important parameter needed for fuel performance evaluation. For the irradiated MOX fuel's Post-Irradiation Examination, the ^{148}Nd method is used to measure the burnup. The fission product ^{148}Nd is an ideal burnup indicator, when appropriate correction factors are applied. In the ATR test environment, the spectrum-dependent and burnup-dependent correction factors (see Section 5 for detailed discussion) can be substantial in high fuel burnup. The validated Monte Carlo depletion tool (MCWO) used in this study can provide a burnup-dependent correction factor for the reactor parameters, such as capture-to-fission ratios, isotopic concentrations and compositions, fission power, and spectrum in a straightforward fashion. Furthermore, the correlation curve generated by MCWO can be coupled with the ^{239}Pu/Pu ratio measured by a Mass Spectrometer (in the new MCWO-MS method) to obtain a best-estimate MOX fuel burnup.

A Monte Carlo - MCWO method can eliminate the generation of few-group cross sections. The MCWO depletion tool can analyze the detailed spatial and spectral self-shielding effects in UO_2, WG-MOX, and reactor-grade mixed oxide (RG-MOX) fuel pins. The MCWO-MS tool only needs the MS-measured ^{239}Pu/Pu ratio, without the measured isotope ^{148}Nd concentration data, to determine the burnup accurately. MCWO-MS not only provided linear heat generation rate, Pu isotopic composition versus burnup, and burnup distributions within the WG-MOX fuel capsules, but also correctly pointed out the inconsistency in the large difference in burnups obtained by the ^{148}Nd method.

1. INTRODUCTION

For the disposition of surplus weapons-grade plutonium (WG-Pu) via light water reactors (LWRs), it is important to demonstrate that the available experience base with Reactor-Grade Mixed uranium-plutonium OXide (RG-

MOX), which was generated primarily in Europe, can be applied to WG-MOX in LWRs. To support potential licensing of MOX fuel made from WG-plutonium and depleted uranium for use in United States reactors, an experiment containing WG-MOX fuel has been fabricated and is being irradiated in the Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL). The uninstrumented test assembly containing nine MOX fuel capsules and neutron monitor wires was inserted into the ATR for irradiation to achieve a burnup of 50 GWd/t. The Oak Ridge National Laboratory (ORNL) manages this project for the Department of Energy.

The Post-Irradiation Examination (PIE) being conducted by ORNL uses the ^{148}Nd method[1] to measure the fuel burnup, which is one of the most important parameters characterizing WG-MOX performance. The fission product ^{148}Nd is a highly reliable burnup measurement indicator, when appropriate correction factors are applied. In the ATR test environment, the spectrum-dependent correction factors can be substantial in high fuel burnup. The Monte Carlo depletion tool (MCWO)[2] used in this study can provide an accurate correction factor for the reactor parameters, such as capture-to-fission ratios, isotopic concentrations and compositions, fission power, and spectrum versus burnup.

Furthermore, the correlation curve generated by <u>MCWO</u> can be coupled with the ^{239}Pu/Pu ratio measured by a <u>M</u>ass <u>S</u>pectrometer. This synthesis results in the MCWO-MS method developed for the work reported here, which gives a best-estimate MOX fuel burnup.

2. WG-MOX FUEL TESTING IN THE ATR

The initial experiment phase (Phase-I irradiation), which contained nine MOX fuel capsules, was loaded into the ATR in January 1998. After 153.5 effective full power days (EFPDs) of irradiation in Phase-I,[3] a capsule pair was withdrawn from the ATR in September 1998 after having achieved an average discharge burnup of about 8.6 GWd/t. At the end of Phase-II[4] irradiation (226.9 EFPDs), an additional capsule pair was withdrawn in September 1999 after having achieved an average discharge burnup of about 21.5 GWd/t. Also, at the end of Phase-III[5] irradiation (232.8 EFPDs), an additional capsule pair was withdrawn in September 2000, after having achieved an average discharge burnup of about 29.6 GWd/t. Post-Irradiation Examination (PIE) of these capsules has recently been completed at ORNL The maximum burnup to be achieved in this test was originally set at 30 GWd/t. It was subsequently decided that the WG-MOX fuel would be irradiated to a burnup of 50 GWd/t. Future PIEs will involve a capsule pair to be withdrawn in March 2002 at 40 GWd/t and three capsules to be withdrawn in October 2003 at 50 GWd/t.

3. MCWO-MS METHOD

One of the most important parameters needed for fuel performance evaluation is the burnup of the irradiated WG-MOX fuel pellets. For the MOX fuel's PIE, ORNL uses the ^{148}Nd method to measure the fuel burnup. The fission product ^{148}Nd is an ideal burnup indicator. However, the transmutation of ^{147}Nd to ^{148}Nd will introduce a systematic error whose contribution must be corrected. Furthermore, the higher ^{241}Pu wt% in the WG-MOX fuel will produce about 18% more 148-mass-chain fission yield[1] than the ^{239}Pu and ^{235}U per fission, whose contribution must also be corrected. These corrections are relatively small in LWRs and can be treated routinely. In an ATR irradiation, where thermal neutron fluxes are high, this correction can be substantial in the high fuel burnup region.

Mass Spectrometry (MS) in the ^{148}Nd method can be calibrated to achieve a highly accurate measurement by eliminating the mass discrimination bias. Mass ratios can be obtained by MS with a precision of about 1%. The ^{148}Nd method can provide additional information about the uranium and plutonium concentrations and isotopic compositions on the same sample taken for the burnup analysis. The determination of burnup from changes in isotopic composition[6] needs an accurate correction factor of the capture-to-fission ratios for ^{235}U, ^{239}Pu, and ^{241}Pu versus burnup. In the following, we will describe a new Monte Carlo depletion tool that can provide the needed correction factor for the reactor parameters.

In general, reactor physics analysis consists of multistep analysis methods. The multigroup diffusion equation with node-wise constant cross sections requires the fuel assembly to be appropriately homogenized. However, the complex spectral transitions in the WG-MOX fuel pellet present a serious challenge. The major source of uncertainty in the fuel burnup calculation comes from burnup-dependent cross-section (XS) and resonance treatment of neutron fluxes in the MOX fuel pellet. To avoid these problems, a validated depletion tool is used, which applies the Monte-Carlo code MCNP,[7] coupled with an isotope depletion code, ORIGEN-2[8]; this is the MCWO[2,9] methodology. MCWO was used to analyze the fission power density ratio and cumulative burnup of MOX fuel pellets versus irradiation effective full power days (EFPDs). MCWO can provide an accurate correction factor of the reactor parameters, such as capture-to-fission ratios, isotopic concentrations and compositions, fission power and spectrum versus burnup. As a result, the correlation curve generated by MCWO can be coupled with the ^{239}Pu/Pu ratio measured by the Mass Spectrometer (in the new MCWO-MS method) to obtain a best-estimate MOX fuel burnup.

MCWO was used to track fuel burnup and heat rates as functions of irradiation time. The fission power distribution and linear heat generation rate (LHGR) of the MOX fuel capsules were provided for the thermal analysis. Temperature distributions were needed to make sure that the MOX pins met the ATR safety requirements and to analyze the behavior of the fission gas release. The MCWO-calculated results were also provided to ORNL for the experiment-specific Capsule Assembly Response Thermal and Swelling (CARTS) code fuel performance analysis.

In the ATR test environment, the total heating rate in the MOX fuel pellet is the sum of the neutron, prompt gamma, and fission products delayed gamma heating (in the fuel pellet and coming from the ATR core), which is a rather complicated process. However, the fission heating tally (F7) in MCNP assumes all the prompt gamma heat generated from the fission in the fuel pellet is deposited locally without any leakage, which can compensate for the incoming gamma heating from ATR core. Furthermore, the total heating rate is dominated by fission heat (about 96%) in the fuel pellet. So, to simplify the as-run physics analysis, only the fission heat tally F7 was chosen for the total heat rate (LHGR) and burnup (LHGR-estimated burnup) conversion.

A good indicator of fuel burnup is the Fissions per Initial heavy Metal Atom (FIMA). This is simply the ratio of the number of fissions that have occurred in the fuel to the initial (zero burnup) inventory of heavy metal atoms (uranium plus plutonium) in the fuel. A FIMA value is determined as part of the normal PIE burnup determination procedure. Based on an effective energy release of 205.4 MeV per fission, burnup (GWd/t) is then obtained by multiplying the FIMA (%) by the conversion factor 9.60. To improve the clarity of information, in addition to the burnups corresponding to the calculated LHGRs (LHGR-estimated burnups), this paper will use the MCWO-calculated FIMA results to convert to burnups (GWd/t).

4. WG-MOX FUEL TESTING ASSEMBLY MODEL

MCNP can model extremely complex three-dimensional geometries. MCWO is quite accurate over a given region because MCNP-generated reaction rates are integrated over the continuous-energy nuclear data and the space within the region. Thus, any oddly or regularly shaped region in MCNP can be depleted (on average). Applying this capability allows calculation of detailed nuclide concentration and power distributions within the MOX capsule as functions of burnup. Its disadvantage is a longer computational time to achieve the required tally precision and minimize statistical fluctuations in the results.

There are three MOX fuel test sections axially, with the center section at the core midplane, and three fuel capsules in each section, for a total of nine fuel capsules in the test assembly, which were all included in the ATR MCNP Core Model (ATRM) as shown in Fig. 1. The WG-MOX test fuel pellet comprises five percent PuO_2 and 95%

depleted UO_2. Each fuel capsule is 0.415 cm in radius and 15.24 cm in length and contains 15 MOX fuel pellets. Channel 1 capsules are located away from the ATR core center, behind the capsules in channels 2 and 3. The adjacent flux-wire channel X is closer to the core center, in front of the flux wires in channels Y and Z as shown in Fig. 2. The MCNP-calculated thermal and fast neutron fluxes are benchmarked with measured Co-59 (thermal neutron flux) and Ni-58 (fast neutron flux) neutron monitor data. In Ref. 10, the averaged thermal neutron flux Calculated/Measured (C/M) ratios of channels X, Y, and Z are 1.05, 1.08, and 1.00, respectively, which is good agreement for this experiment.

For a small MOX fuel test assembly, the MCNP Monte Carlo code would spend a lot of computer time tracking neutrons in the surrounding medium. A detailed ATRM run requires 822 minutes of IRIX64 workstation computer time to achieve one relative standard deviation (1σ) less than 1.50% in the fuel capsule fission tallies. To reduce computer time, a new isolated box model with boundary source (IBMBS) model[11] was developed. To generate an ATR boundary source from the MCNP model, an I-hole cylindrical shell, as shown in Fig. 2, was placed around the test assembly in the ATRM. For a typical MCNP ATRM, the total volume is 1.43 x 10^6 cm^3, and the volume of the box is 3.66 x 10^3 cm^3. The ratio of the quarter core to box volume ratio is 390.7. For typical ATRM and IBMBS MCNP calculations, the CPU times to achieve the same one relative standard deviation (1.5%) are 490 and 3.5 minutes, respectively. As a result, the efficiency of the cell tally has increased by about 490/3.5=140.0 times for the IBMBS.

MCNP (with the Surface Source Write - SSW option) was used with ATRM to generate the box shell boundary neutron source (BSBNS) file. The generated BSBNS file was then used in the Isolated Test Assembly Model (with the Surface Source Read - SSR option in MCNP) to calculate the burnup-dependent cross section and fission rate distributions. The results show that MCWO with the ATR boundary neutron source[11] can accurately handle the neutron space and energy resonance interaction and generate burnup-dependent XS for the Pu and U isotopes in the fuel pellet. The validated MCWO method was used to perform the neutronics analysis of WG-MOX fuel in the ATR. The prediction of nuclide profiles and burnup distributions in irradiated MOX fuel pellets via this new methodology can provide valuable data for MOX fuel performance evaluation.

5. RESULTS AND DISCUSSION

The experimental results of the Average Power Test (APT) include observations from the fuel fabrication process, PIE findings, U and Pu isotopic composition, and MOX fuel burnup.[3,4,5] All of the capsules were visually examined in the transfer canal at the ATR during the shuffling and transfer to ORNL for post-irradiation examination (PIE). All of the irradiated capsules appeared as fresh as they did at the original insertion. No changes in the external dimensions were noted. Oxidation of the external surfaces was likewise not noticeable. No appreciable scratches or wear spots were observed as might occur from fretting. MCWO was used to track fuel burnup and heat rates as functions of irradiation time. In summary, no anomalous indications were seen.

5.1 MCWO-CALCULATED ^{241}PU, ^{238}U (FAST) AND ^{147}ND CORRECTION FACTOR VERSUS FIMA BURNUP IN ATR

The transmutation of ^{147}Nd to ^{148}Nd will introduce a systematic error in the ^{148}Nd method, and results need to be corrected. In addition, the higher ^{241}Pu wt% in the WG-MOX fuel irradiation will produce about 18% more of 148-mass-chain fission yield[1] than the ^{239}Pu and ^{235}U per fission, whose contribution must also be corrected. The sums of the 148-mass-chain yield, which consists of ^{148}Cs, ^{148}Ba, ^{148}La, ^{148}Ce, ^{148}Pr, and ^{148}Nd yields, of the ^{235}U, ^{239}Pu, and ^{241}Pu, are 0.0167312 ± 0.35%, 0.016422 ± 0.5%, and 0.0193209 ± 0.7%, respectively.[1] For the fast fission of ^{238}U, the 148-mass-chain yield is 0.0209416 ± 1.0%.[1] All the ^{148}Cs, ^{148}Ba, ^{148}La, ^{148}Ce, and ^{148}Pr isotopes in the 148-masschain have rather short half-lives, which will β-decay to the stable isotope ^{148}Nd at the end of the decay chain. These 148-mass-chains indicate that ^{241}Pu in the WG-MOX fuel will produce about 18% more ^{148}Nd fission yield[1] than the ^{239}Pu and ^{235}U per fission. These corrections are relatively small in LWRs and can be treated routinely. In an ATR irradiation, where thermal neutron fluxes are high, this correction can be relatively significant in the high fuel burnup region.

To calculate the ^{147}Nd, ^{238}U and ^{241}Pu total correction factor for the ^{148}Nd method versus burnup, first MCWO was used to calculate the depletion/build-up of the ^{148}Nd(A) in the irradiated MOX fuel pellet versus the EFPDs (from MOX fuel irradiation Phase-I to Phase-IV part 2). Second, the capture cross section of the ^{147}Nd was set to zero and the ^{238}U and ^{241}Pu fission yield distribution was assumed the same as that of ^{239}Pu in the ORIGEN-2 cross-section library. Then, the same MCWO-calculation procedure as in step 1 was performed to calculate the depletion/buildup of ^{148}Nd(B) versus the EFPDs. The MCWO-calculated ^{147}Nd and ^{241}Pu total correction factor versus burnup is obtained by ^{148}Nd(B) / ^{148}Nd(A) and plotted in Fig. 3. Figure 3 shows that the total correction factor decreases from 1.00 at BOL to 0.97 at the burnup of 50 GWd/t.

5.2 DETERMINATION OF THE MOX FUEL BURNUP BY THE MCWO-MS METHOD

All the withdrawn capsule pairs in the MOX fuel test assembly had the same initial ^{239}Pu weight percent (93.81%). This decreases monotonically but not linearly with burnup. FIMA MCWO-calculated and as-run MCNP-calculated[9] ratios of ^{239}Pu/Pu are shown versus burnup in Fig. 4. The as-run FIMA MCWO-calculated ^{239}Pu/Pu ratios and burnups at the end of Phase I Phase II, and Phase III are 84.00%, 9.2 GWd/t; 67.00%, 21.2 GWd/t; and 45.30%, 30.1 GWd/t; respectively. The as-run FIMA MCWO-calculated ^{239}Pu/Pu ratios and burnups have excellent agreement with the correlation curve of the ^{239}Pu/Pu ratio and burnup as shown in Fig. 4.

The as-run MCNP-calculated ^{239}Pu/Pu ratios and the LHGR-estimated burnup at the end of Phase I, Phase II, and Phase III are 83.6%, 8.6 GWd/t; 63.5%, 21.5 GWd/t; and 43.2%, 29.6 GWd/t; respectively, which agree quite well with the as-run FIMA MCWO-calculated burnups (within $1\sigma=\pm3.0\%$).

5.3 RESOLUTION OF THE DISCREPANCIES BETWEEN THE 148ND-MEASURED AND MCWO-CALCULATED BURNUP

The paired capsules 3 and 10 withdrawn at the end of the Phase-III irradiation had the same initial ^{239}Pu weight percent (93.81%). The measured values of the ^{239}Pu/Pu ratio at the 30-GWd/t withdrawal are 50.34% and 49.45%, respectively.[5] These data strongly indicate that the two burnups differ by no more than about two percent. Using the newly developed MCWO-MS method in this study, a confirmatory burnup analysis for Capsules 3 and 10 has been performed.

The ^{148}Nd-Mass-Spectrometer-measured (^{147}Nd, ^{238}U, and ^{241}Pu corrected) burnups and the ^{239}Pu/Pu ratios at the end of Phase I, II, and III (with mark ■) are plotted in Fig 5. Based on the correlation curve between ^{239}Pu/Pu ratio and burnup in Fig. 4 and the MS measured ^{239}Pu/Pu ratio, the values of burnup best estimated by interpolating the polynomial regression curve fitting for Capsules 3 and 10 are 29.33 (^{239}Pu/Pu = 50.34%) and 29.76 (^{239}Pu/Pu = 49.45%) GWd/t, respectively. The figure 5 shows an excellent agreement with the ^{48}Nd-Mass-Spectrometer-measured data at the burnups of Phase I and II, but exhibits some deviation at the burnup near 30 GWd/t for Phase III. However, at the end of Phase III irradiation, the as-run LHGR-estimated burnup of 29.6 GWd/t[9] agrees quite well with the FIMA MCWO-MS-measured burnup of 29.33 and 29.76 GWd/t.

The paired WG-MOX fuel Capsules 3 and 10 occupied the same symmetric test assembly positions throughout their irradiation in ATR. The burnup values of Capsules 3 and 10 should be close to each other. MCWO-MS predicts burnups of 29.33 and 29.76 GWd/t by interpolating the polynomial regression fitting ^{239}Pu/Pu correlation curve, which is consistent with the measurement of the ^{239}Pu/Pu ratio in showing a burnup variation of less than 2 percent. However, the ^{148}Nd burnup measurements taken for the PIE indicate 32.9 GWd/t for Capsule 10 (front upper left) and 26.9 GWd/t for Capsule 3 (front upper right), which represents a 22% difference. In thermal reactors like the ATR and PWR, the fission reactions are mostly caused by thermal neutrons. The estimated uncertainties in the thermal (2200 m/s) and fast (E > 1 MeV) neutron fluxes in the ATR, at the 68 percent confidence level, are ± 3% and ± 5%, respectively. The measured thermal neutron fluences at the symmetric (left and right) flux wire "Z" and "Y" (see Fig. 2) positions show a variation of 2.3% ($1\sigma = \pm 3.0\%$). Although the measured fast neutron fluence shows a larger variation of 5.6% ($1\sigma = \pm 5.0\%$), the 22% difference as derived from the paired-capsule burnup measurement using the ^{148}Nd method fails a consistency check.

Upon review of the Radioactive Materials Analysis Laboratory (RMAL) analytical procedure, an inaccuracy has been identified in one step that significantly affects one of the measured burnups initially reported. The cause of the inaccuracy has been determined to lie in the neodymium content as determined by the Thermal Ionization Mass Spectrometer (TIMS). A correction factor has been applied to the interpretation of the neodymium readings for both pellet samples. After applied ^{148}Nd correction factor (0.988), the revised burnups obtained by the neodymium method are 27.27 and 27.47 GWd/t (without final validation and verification), respectively, with a one standard uncertainty band of ±5.0%[12]. Based on the FIMA MCWO-calculated ^{239}Pu/Pu correlation versus burnup curve in Fig. 4, the Capsules 3 and 10, MCWO-MS method produces the revised burnups of 29.4 (^{239}Pu/Pu = 50.21%) and 29.8 (^{239}Pu/Pu = 49.32%) GWd/t, respectively, and plotted in Fig. 5 with mark ★. These results agree within 8.3%. However, the newly developed MCWO-MS correctly pointed out the inconsistency in the large difference in burnups obtained by the ^{148}Nd method in the pre-revised PIE measurement result.

CONCLUSIONS

A simple, uninstrumented test assembly containing WG-MOX fuel capsules was inserted into the ATR. MCWO was used to perform the neutronics analysis for this fuel testing in the ATR. Important neutronics parameters were computed using MCWO methods. These computations led to an experiment design for a WG-MOX fuel assembly that met all safety design requirements. Considering the complicated ATR geometry and the uncertainty of the core power distribution in each lobe, it is remarkable that the results matched so well with the burnup measurement.

A Monte Carlo - MCWO method can eliminate the generation of few-group cross sections. The MCWO depletion tool can analyze the detailed spatial and spectral self-shielding effects in UO_2, WG-MOX, and reactor-grade mixed oxide (RG-MOX) fuel pins. Any odd or regular shaped region in MCNP can be depleted (on average) with reaction rate data that can be more accurate than the few group data used in the commercial LWR industry. The MCWO method can calculate the burnup-dependent reactor parameters, such as capture-to-fission ratios, isotopic concentrations and compositions, fission power, and spectrum in a straightforward fashion and treat the entire fuel assembly at once. In addition, the MCWO-MS method only needs the MS-measured ^{239}Pu/Pu ratio, without the measured isotope ^{148}Nd concentration data, to determine the burnup accurately.

This is significant because the newly developed MCWO-MS method not only provided LHGR, Pu isotopic composition versus burnup and burnup distributions within the WG-MOX fuel capsules, but also correctly pointed out the inconsistency in the large difference in burnups obtained by the ^{148}Nd method. As a result, it leads to an review of the RMAL analytical procedure, and revised the burnups obtained by the neodymium method to 27.27 and 27.47 GWd/t, respectively. The MCWO-MS method can also be used in a wide variety of other applications, including advanced fuel cycle (Advanced High Temperature gas-cooled Reactors) performance analysis, long life minor actinide transmutation, strong absorber depletion analysis, MOX fuel and reactor materials test assembly design, and ATR test as-run physics analysis.

REFERENCES

1. Standard Test Method for Atom Percent Fission in Uranium and Plutonium Fuel (Neodymium-148 Method)", *ASTM Standards*, E-321-96.
2. G. S. Chang And J. M. Ryskamp, "Depletion Analysis of Mixed Oxide Fuel Pins in Light Water Reactors and the Advanced Test Reactor," *Nucl. Technol*., **Vol. 129, No. 3**, pp. 326-337 (2000).

3. R. N. Morris, C. A. Baldwin, B. S. Cowell, S. A. Hodge, et. al. "*MOX Average Power Early PIE: 8 GWd/MT Final Report,*" Oak Ridge National Laboratory, ORNL/MD/LTR-172, (November 1999).
4. R. N. Morris, C. A. Baldwin, S. A. Hodge, L. J. Ott, C. M. Malone, N. H. Packan, "*MOX Average Power Intermediate PIE: 21 GWd/MT Final Report,*" Oak Ridge National Laboratory, ORNL/MD/LTR-199, (December 2000).
5. R.N. Morris, C. A. Baldwin, S. A. Hodge, N. H. Packan, "*MOX Average Power 30 GWd/MT PIE: Final Report*," Oak Ridge National Laboratory, ORNL/MD/LTR-212, (November 2001).
6. Standard Test Method for Atom Percent Fission in Uranium and Plutonium Fuel (Mass Spectrometric Method), *ASTM Standards*, E-244-80, (Re-approved 1995).
7. J. Briesmeister (Editor), "MCNP—A General Monte Carlo N-Particle Transport Code", **Version 4B, LA-12625-M, Rev. 2**, Los Alamos National Laboratory (1993).
8. A. G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," *Nuclear Technology,* **Vol. 62**, pp. 335-352 (1983).
9. G. S. Chang, R. C. Pedersen, "Weapons-Grade MOX Fuel Burnup Validation in ATR," *Trans. Am. Nucl. Soc*., **Vol 84,** pp. 239, 2001.
10. G. S. Chang, JW Rogers, and J. M. Ryskamp, "Neutron Flux and Spectrum Variation in a MOX Fuel Experiment," *Trans. Am. Nucl. Soc*., **Vol. 81**, 284 (1999).
11. G. S. Chang and J. M. Ryskamp, "Monte-Carlo Boundary Source Approach in MOX Fuel Test Capsule Design," Trans. *Am. Nucl. Soc*., **Vol. 80**, pp. 268-269 (1999).
12. S. A. Hodge, L. J. Ott, F. P. Griffin, C. R. Luttrell, "*Implications of the PIE for the 30 GWd/MT-Withdrawal MOX Capsules*," Oak Ridge National Laboratory, ORNL/MD/LTR-212**, Vol. 2, Rev. 1,** (February 2002).

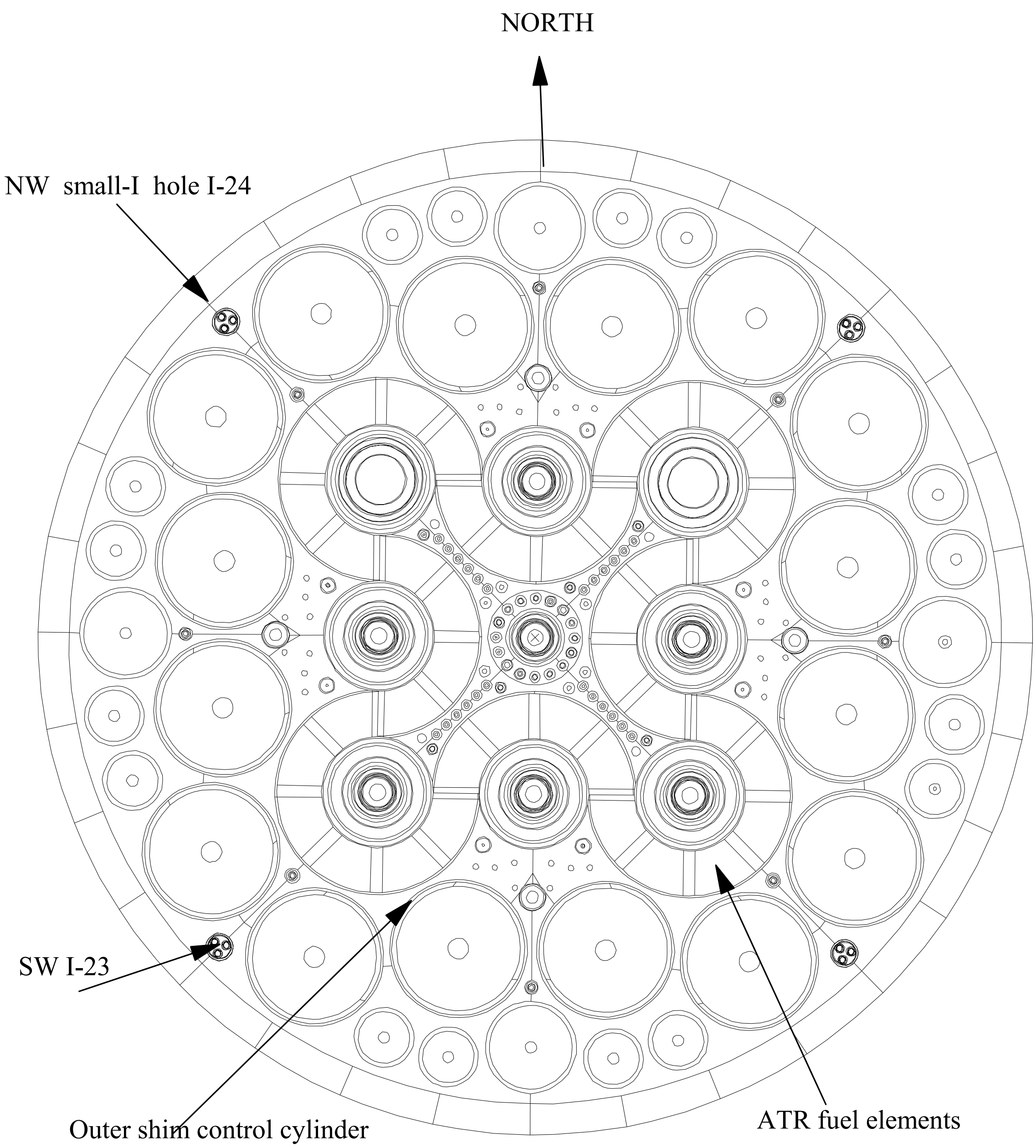

Figure 1: Radial cross-sectional view of the full core MCNP model with two test assemblies at NW and SW small I-hole positions

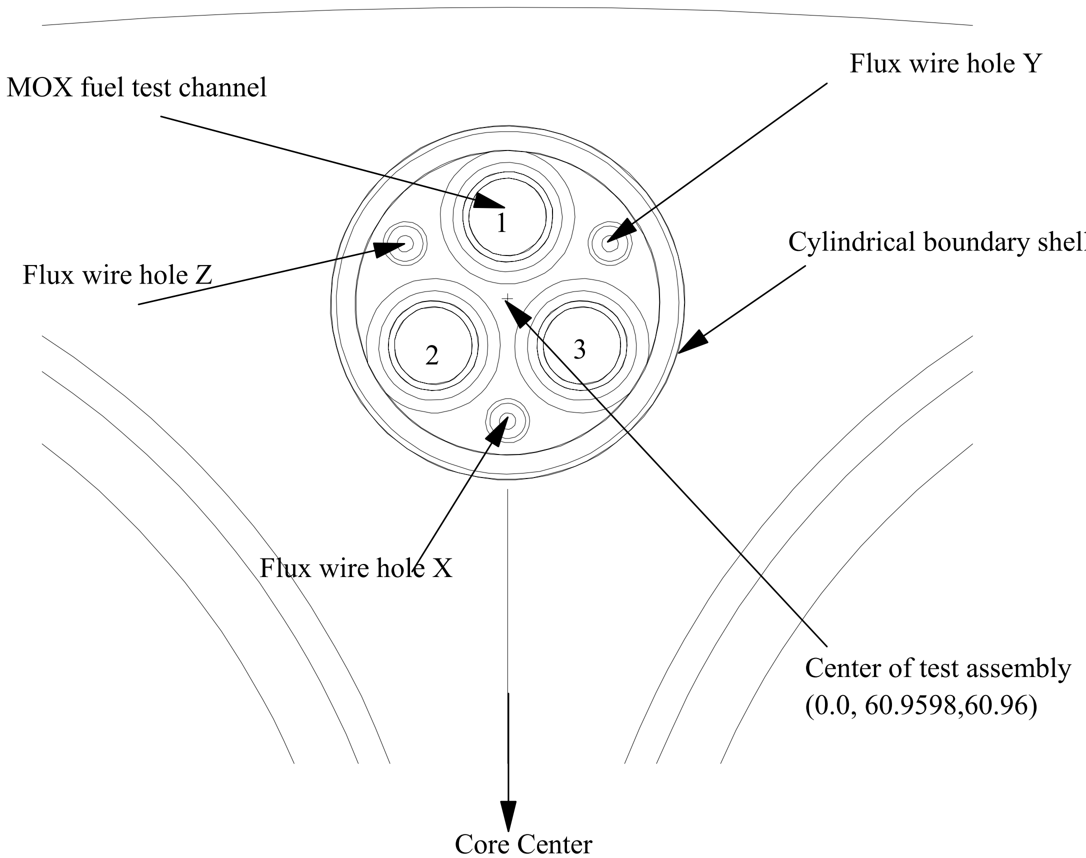

Figure 2: Detailed radial cross-sectional view of the WG-MOX fuel test assembly

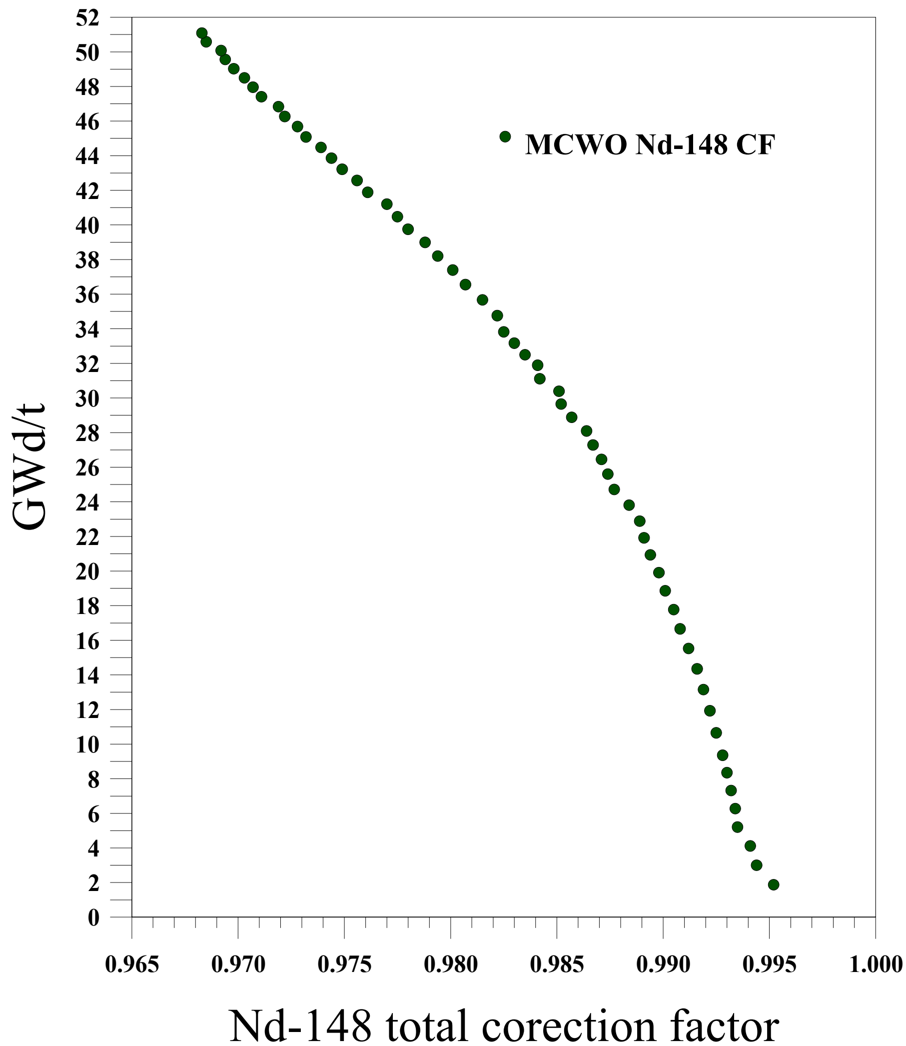

Figure 3. MCWO-calculated ^{147}Nd, ^{238}U (fast), and ^{241}Pu total correction factor for ^{148}Nd method versus burnup

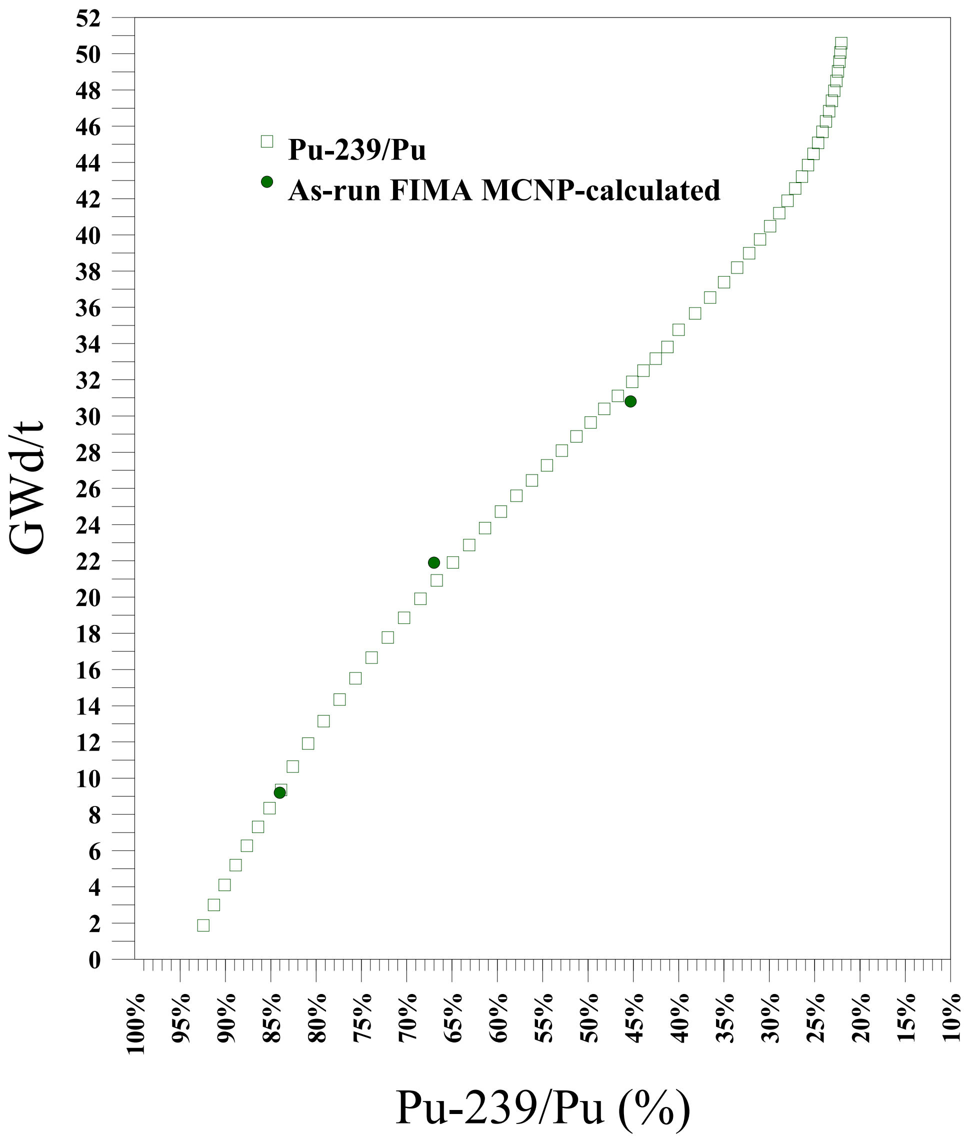

Figure 4. MCWO-MS-calculated and As-run MCNP-calculated ^{239}Pu/Pu ratio versus burnup

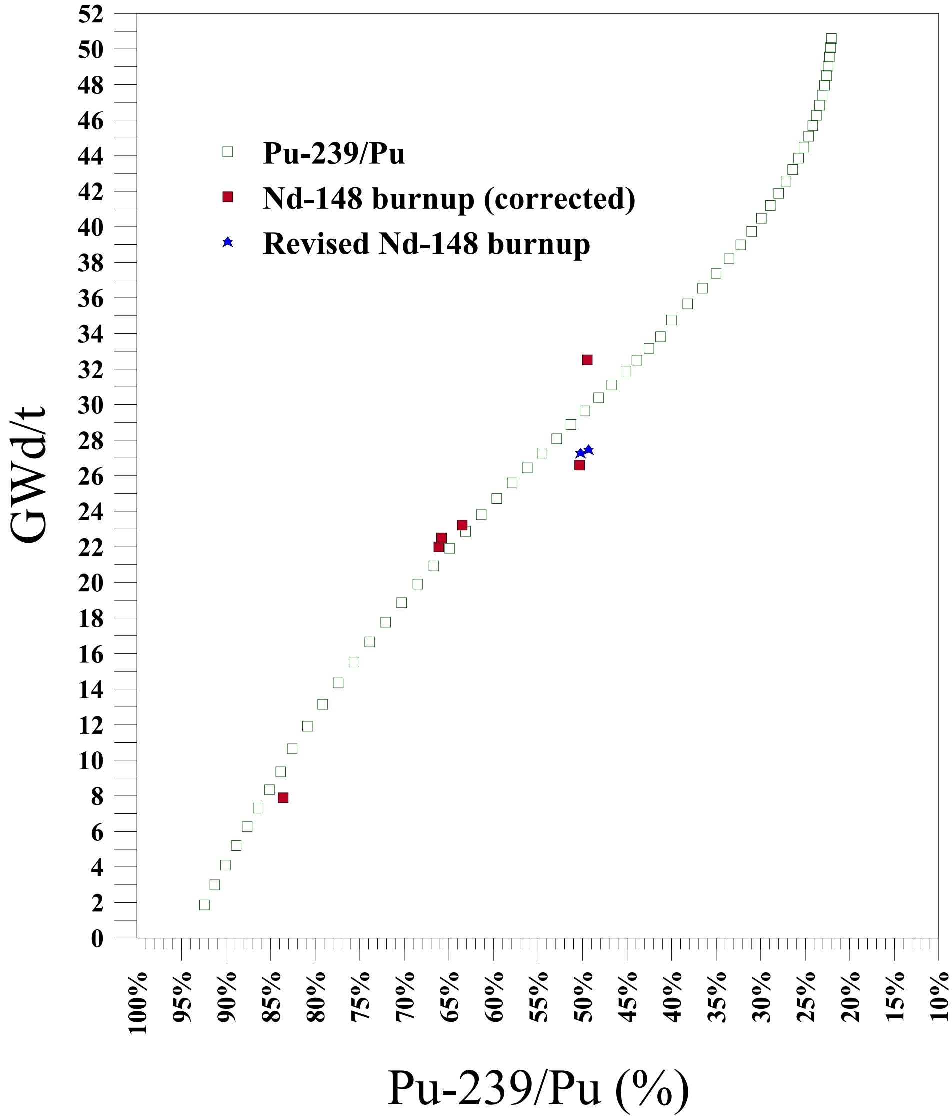

Figure 5. FIMA MCWO-calculated and Mass-Spectrometer-measured ^{239}Pu/Pu ratio versus burnup